High Fusion Performance in Super H-Mode Experiments on Alcator C-Mod and DIII-D

P. B. Snyder1, J. W. Hughes2, T. H. Osborne1, C. Paz-Soldan1, W. Solomon1, D. Eldon1, T. E. Evans1, T. Golfinopoulos2, R. J. Groebner1, A. E. Hubbard2, M. Knolker3, B. LaBombard2, F. M. Laggner3, O. Meneghini1, S. Mordijck4, S. Scott3, H. R. Wilson5, and Y. B. Zhu6

1General Atomics, San Diego, CA 92186, USA
2Plasma Science & Fusion Center, MIT, Cambridge, MA 02139, USA
3Princeton Plasma Physics Laboratory (PPPL), Princeton, NJ 08540, USA
4College of William & Mary, Williamsburg, VA 23185, USA
5University of York, Heslington, UK
6University of California Irvine, CA 92697, USA

Corresponding Author: P. B. Snyder, snyder@fusion.gat.com

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General Atomics
San Diego CA, USA
Email: snyder@fusion.gat.com

J.W. HUGHES, T. GOLFINOPOULOS, A. HUBBARD, B. LaBOMBARD
MIT Plasma Science and Fusion Center
Cambridge MA, USA

M. KNOLKER, B. GRIERSON, F. LAGGNER, S. SCOTT
Princeton University / Princeton Plasma Physics Laboratory
Princeton NJ, USA

S. MORDIJCK
College of William and Mary
Williamsburg VA, USA

H.R. WILSON
York Plasma Institute, University of York
York, UK

Y.B. ZHU
University of California-Irvine
Irvine CA, USA

Abstract

The “Super H-Mode” regime is predicted to enable pedestal height and fusion performance substantially higher than for standard H-Mode operation. This regime exists due to a bifurcation of the pedestal pressure, as a function of density, that is predicted by the EPED model to occur in strongly shaped plasmas above a critical pedestal density. Experiments on Alcator C-Mod and DIII-D have achieved access to the Super H-Mode (and Near Super H) regime, and obtained very high pedestal pressure, including the highest pedestal pressure ever achieved on a tokamak (p_{ped} \approx 80 kPa) in C-Mod experiments operating near the ITER magnetic field. DIII-D Super H experiments have demonstrated high performance, including the highest stored energy in the present configuration of DIII-D (W \approx 2.2-3.2 MJ), while utilizing only about half of the available heating power (P_{heat} \approx 7-12 MW). These DIII-D experiments have achieved the highest value of peak fusion gain, Q_{DT,equiv} \approx 0.5, ever achieved on a medium scale (R<2m) tokamak. Sustained, stationary high performance has been maintained in the presence of deuterium and nitrogen gas puffing, which enables a more radiative divertor condition. Super H-Mode access is predicted for ITER and expected, based on both theoretical prediction and observed normalized performance, to enable ITER to achieve its performance goals (Q=10) at I_p < 15MA, and to enable more compact, cost effective DEMO designs.

1. INTRODUCTION – THE EPED MODEL AND PREDICTION OF THE SUPER H REGIME

Developing a high performance, consistent core-edge plasma solution is key to the development of cost-effective magnetic fusion power. In a tokamak, the edge transport barrier, or “pedestal,” region plays a critical role in this dynamic. The pressure at the top of the pedestal, or “pedestal height,” strongly impacts global confinement and fusion performance, with fusion power production expected to scale approximately with the square of the pedestal height. The density at the interface between the pedestal and open field line region, i.e. the separatrix density, must be consistent with a divertor solution which enables low temperatures and minimal erosion at material surfaces.

The EPED model [1–4] predicts the H-mode pedestal height and width based upon criticality to two calculable constraints: 1) onset of non-local peeling-ballooning (P-B) modes, 2) onset of nearly local kinetic ballooning modes (KBM). These calculations are performed on realistic model equilibria, with self-consistent bootstrap current in the pedestal region, to enable pedestal predictions for future experiments and future devices. The combination of P-B and KBM physics leads to strong dependencies of the pedestal height on poloidal field (B_p), toroidal field (B_t) and plasma shape, which have been successfully tested in several experiments [eg 1-10].
important dependence on density (which is an input to EPED) derives primarily from the variation of the bootstrap current with collisionality. In most circumstances, the EPED model predicts a single pedestal solution, at the intersection of P-B and KBM criticality (see Fig. 1a). However, for very strongly shaped plasmas (typically triangularity $\delta > -0.45$, for $q_{95} \sim 3.5-4.5$, $R/a \sim 3$, elongation $\sim 1.7-2$), above a critical density, the solution can bifurcate into three solutions. In this circumstance we denote the lower pressure solution the “H-mode” solution, and the two higher pressure solutions “Super H-mode” (SH) solutions because they sit above the H-mode solution in pressure (at the same set of EPED input parameters, including density, $Z_{eff}$, shape, current, field, and major and minor radius). The Super H-mode solutions may initially appear to be of limited practical interest, because they sit above a P-B unstable region, and P-B modes are known to drive either edge localized modes (ELMs) or edge harmonic oscillations (EHOs) which prevent further increases in pressure. However, diagrams such as Fig. 1a are drawn for a fixed set of input parameters. By varying key parameters, for example by reducing the pedestal density, it is possible to shrink the P-B unstable region between the H and SH solutions until it disappears, leaving only a single (what was the highest pressure) solution. After the pedestal rises to this solution, the pedestal density can then be increased, leaving the discharge in Super H-mode. This sort of parametric trajectory can be visualized (magenta arrow) in Fig. 1b, starting at high pedestal density (along black line at the right), decreasing the density until only the blue solution remains, then increasing density to access the red SH solutions. Note that the blue “Near Super H” (NSH) regime shown in Fig 1b also results higher pedestal pressure than for typical H-modes (black lines). Coupled core-pedestal modeling [11-14] predicts high fusion performance should be obtainable in both the NSH and SH regimes.

Accessing the Super H regime experimentally requires a strongly shaped plasma at moderate $q_{95} \sim 3.5-5$, and an appropriate parametric trajectory for the density. The Super H regime was first discovered experimentally on DIII-D, in a series of experiments guided by prior theoretical predictions [15,16]. These initial experiments were conducted with neutral beam injection in the counter-current direction, and generally had a quiescent (QH mode) edge, which enabled a smooth increase of the density in the Super H regime. Additional DIII-D experiments and predictive modeling have scoped out the parameter regime for SH access on DIII-D, and the potential benefits for fusion performance [16-18].

In the following sections we describe more recent Super H mode experiments on Alcator C-Mod and on DIII-D (experiments on C-Mod during its final month of operations in September 2016, and experiments on DIII-D, with co-current NBI and ELMs, from June 2017 – April 2018). These experiments were aimed at further testing of EPED model predictions, as well as realization of very high pedestals, and high fusion performance. In Section 2, we describe experiments accessing the Super H regime with very high pedestal and global pressure, including record (~80kPa) pedestal pressure achieved on C-Mod, and high stored energy and fusion gain on DIII-D. In Section 3, we describe progress toward sustainment of SH/NSH discharges on DIII-D, using 3D magnetic perturbations to drive density and impurity transport, and initial study of the compatibility of high core and pedestal performance with strong deuterium and nitrogen puffing in the divertor region.

![Figure 1: (a) Illustration of the EPED model, which predicts pedestal height and width via calculated P-B (blue lines) and KBM (green dashed line) constraints. Most commonly only a single solution (black circle) is found, but for strong shaping above a critical density, this solution bifurcates and two additional solutions (red circles), called “Super H mode” (SH) solutions are found. (b) EPED predictions as a function of density indicate paths to Super H mode (arrows) via reaching sufficiently low pedestal density, and then increasing density over time. The green squares are measurements from DIII-D discharge 171322 at 1.6MA. Good agreement is found between the DIII-D observations and the EPED predictions (thick red, blue and black lines), which indicate access to the Near Super H and Super H regime.](image-url)
2. ACHIEVING SUPER H ACCESS AND HIGH PEAK PERFORMANCE

The high pedestal pressure predicted and observed in the SH and NSH regimes is expected to enable high global pressure and confinement. This is due both to the nature of gradient-scale-length-driven turbulent transport in the core, which causes global pressure to rise with the pedestal, and to the broad pressure profiles enabled by a high pedestal, which have high global MHD limits. Achieving this high level of performance requires both successfully accessing the SH/NSH regime, and avoiding issues such as impurity accumulation and performance-limiting instabilities such as large tearing modes in the core.

Following the prediction and observation of Super H-Mode on DIII-D, predictions were undertaken for additional devices, including Alcator C-Mod and ITER. Predictions were made for Alcator C-Mod by starting with parameters from a high triangularity ($\delta=0.49$) ELMy H-mode (1101214029), and varying the pedestal density over a wide range (Fig 2a). These calculations indicate that the density required to access the SH/NSH regime is substantially lower than typical operating densities on C-Mod (e.g. the green cross in Fig 2a).

Accessing low density presents challenges, particularly in a high-Z metal wall device such as C-Mod, for two reasons (a) the I-H power threshold rises substantially at low density [19,20], (b) impurities can accumulate leading to large radiative losses before ELMs or other edge modes are driven strongly enough to regulate core impurity accumulation. A fortuitous discovery enabled successful access to the required low density regimes. By starting operation in the unfavorable grad-B drift direction (here upper single null), it is possible to first access the I-mode regime [21,22], leading to a relatively low density, high temperature edge plasma, and then transitioning the magnetic balance to the favorable grad-B drift direction (here lower single null), to enter H mode at low density with low impurity content.

Employing this technique, C-Mod discharges at 0.8, 1.0, and 1.4MA were able to access the NSH/SH region [23]. Pedestal densities as low as 7 $10^{19}$ m$^{-3}$ (among the lowest ever achieved on C-Mod) and $T_{ped} > 1.4$ keV were routinely achieved. Discharges with intentionally lowered triangularity exhibited low-n (n=1) edge modes, consistent with encountering the predicted current-driven kink/peeling mode limit (upper blue/red boundary in Figs. 2a,b). Starting at these low densities, the pedestal pressure increased with increasing density, consistent with expectation from theory, and pedestal pressures as high as 70 kPa at 1MA, and 81 kPa at 1.4 MA were achieved. The 81 kPa discharge (1160930042) has the highest pedestal pressure reported on any tokamak, and (Fig 2b) is consistent with access to the Super H regime. The plasma cross section and density and temperature profiles for the highest pedestal pressure case (1160930042) are shown in Fig. 3a. This 1.4MA, 5.8T case had a product of toroidal and poloidal field comparable (~90%) to the ITER values, and it achieved a pedestal pressure roughly 90% of the EPED predicted value for the ITER baseline. These high pedestal pressure discharges on C-Mod enabled comparisons of the EPED model with observations to extend across a factor of 70x on 6 devices, to values very close to the ITER prediction (Fig 2c, green circles are new data from C-Mod Super H-mode experiments). The agreement between the model and observations ($\sigma$=0.22) is similar at low and high pressure, with no indication of significant variation in the level of agreement based on pressure or normalized gyroradius ($\rho_*$) across the studied range.

![Figure 2](image-url)
On DIII-D, a series of experiments have been undertaken in co-injected discharges to explore SH and NSH access and sustainment across a wide range of plasma current ($I_p=1.45-2$ MA) at full field ($B_t=2.0-2.2$ T) and a range of high triangularity ($\delta=0.5-0.7$). These discharges undergo an initial L-H transition at low density, and then the density increases over time to reach the SH/NSH regime (see Fig 1b). Deuterium gas puffing, pumping, and active density control using 3D ($n=3$) magnetic perturbations from the internal coil (“I-coil”) are used to control pedestal density and impurity accumulation. Because these discharges use co-injection (unlike most prior DIII-D Super H experiments which used counter-injection [16-18]), the edge exhibits edge localized modes (ELMs), which in some cases onset during the density rise, and in other cases onset later after high pressure is obtained. These ELMs do not, by themselves, generally limit access to the Super H regime. The 1.6MA case shown in Fig 1b exhibited several ELMs during its rise, and the measurements (green squares) are taken shortly before these ELMs. Co-injection leads to more favorable current profiles in the core, enabling high values of peak ($\beta_N\sim3.9$) and sustained ($\beta_N\sim2.9$) beta.

Experiments at higher current ($I_p=1.8-1.98$ MA) exhibit a similar rise in pedestal pressure with density, reaching $p_{\text{ped}}\sim30$ kPa (Fig 4a). The measured pedestal pressure in Fig 4 is determined by summing the electron pedestal pressure from a modified tanh fit to Thomson scattering data with ion pressure obtained via charge exchange measurements in a single channel near the pedestal top. The observed trajectory follows theoretical expectations (~linear increase in $p_{\text{ped}}$ with $n_{\text{ped}}$) consistent with a rise into the NSH and SH regimes at higher density. As the pedestal pressure rises, these experiments reach high peak stored energy ($W_{\text{MHD}}\sim2.4-3.2$ MJ), the highest values recorded in the present (with existing in-vessel pumps and limiters) configuration of DIII-D. These high values of $W_{\text{MHD}}$ are obtained with modest injected power ($P_{\text{nbi}}\sim8-12$ MW, no RF/EC power, negligible Ohmic power $<0.2$ MW) as shown in Fig 4b, roughly half the available power on DIII-D, consistent with high energy confinement ($\tau_E\sim0.2-0.7$ s, $H_{98}\sim1.6-2.5$) during this time.
The combination of high core ion temperature ($T_{i0}$~14-18 keV), stored energy (2-3.2 MJ) and confinement are favorable for fusion performance, and these discharges exhibit significant DD fusion despite modest values of field ($B_t$=2.1-2.2T) and current ($I_p$=1.6-1.98 MA), and the medium size of DIII-D (a=0.6m, R~1.67m, V~20m$^3$).

In the remainder of this section (Figs 5-6) we consider cases with $T_{i0}$ > 12 keV. Peak DD neutron rates as high as 1.86 \times 10^{16}/s are measured (Fig 5a), via a plastic scintillator with ~15% uncertainty [25]. The roughly linear increase, and strong correlation ($r=0.94$), in neutron rate with volume average pressure times stored energy, $<p>W$, shown in Fig 5a, is consistent with predominantly thermal neutrons (a much weaker correlation, $r=0.34$, is found to NBI power). Analysis with TRANSP predicts total neutron rate (black line) consistent with measurements (blue dashed line), (provided a good match is obtained in W), and indicates that ~2/3 of the neutrons are produced by thermal reactions, while ~1/3 are beam-target and beam-beam reactions (with a small fraction from beam-beam), as shown in Fig 5b.

TRANSP has also been used to assess the equivalent DT fusion performance of these discharges. This is done by fixing a 0.4 ion fraction of tritium (and reducing the D fraction by an equivalent amount), while keeping the carbon impurity fraction unchanged. These simulations find an increase of a factor of ~222 in thermal fusion power going from DD to DT fuel, consistent with prior studies at similar $T_i$ [24]. In a DD plasma, each neutron indicates a reaction in each branch, for a total energy of 7.3 MeV per neutron, and the measured peak neutron rate of 1.86 \times 10^{16}/s corresponds to a peak DD fusion power of 22 kW, and an equivalent DT power ~222x this amount, or 4.8 MW. Because neutral beam power can vary with time (feedback control of $\beta_N$ by varying beam power is employed in some discharges), a causal Gaussian back-average over twice the beam slowing time ($\tau_s\sim40$ ms) is used in calculating the effective neutral beam heating power $P_{nbi}$. DT equivalent fusion gain ($Q_{DT,eq}$), the ratio of DT equivalent power to auxiliary power, has been defined in various ways in the literature. We first employ the simple definition used by [24], $Q_{DT,eq}=P_{nbi,DT,eq}/P_{nbi}$. $P_{nbi}$ is the neutral beam power, with causal Gaussian back average over 2$\tau_s$~80ms (Ohmic power is small, < 0.4MW, and no other auxiliary power is used here, as in [24], and only cases where $\dd W/\dd t$ is positive or nearly zero, $\dd W/\dd t > -0.2$MW, are included). $Q_{DT,eq}$ is plotted as a function of $<p>W/P_{nbi}$ in Fig. 6a. Values as high as 0.54 are achieved (max 0.45 for cases...
with fixed $P_{\text{tot}}$. For comparison, the prior highest peak $Q_{\text{DT,eq}}$ value (~0.32) from DIII-D 89776 is plotted (black X) in Fig 6a (along with a similar shot, 89937, as the red X). Note that alternate definitions of Q have been employed in the literature. If $P_{\text{tot}}$ is replaced by $P_{\text{NBI}}=dW/dt$, then a maximum value of $P_{\text{NBI,DT,eq}}/(P_{\text{NBI}}-dW/dt)$ ~1 is found (in DIII-D 174791). If the definition used by JT-60U [26] is employed, where fusion power is divided into thermal ($P_{\text{th,DT,eq}}$) and beam-driven ($P_{\text{NBI,DT,eq}}$) components, with $Q_{\text{DT,eq}}=P_{\text{th,DT,eq}}/P_{\text{th}}+P_{\text{NBI,DT,eq}}/(P_{\text{NBI}}-dW/dt)$ the maximum value in the DIII-D SH/NSH dataset is $Q_{\text{DT,eq}}$ ~0.9. Peak values of $<p>/T_e$ ~65 kPa s, and $n_0 T_e$ ~5.6 $10^{20}$ m$^{-3}$ keV s are achieved. Note that in the high $Q_{\text{DT,eq}}$ ~0.3 DIII-D cases shown here, the high Q condition is maintained only briefly (~0.1-0.4s), as is the case in prior high Q DIII-D cases [24] and reported high Q cases on other devices [26,27]. The important issue of sustainment of high performance discharges is discussed in the following section.

3. SUSTAINMENT AND CORE-EDGE COMPATIBILITY

High pedestal pressure and peak performance indicate the potential of the Super H and Near-Super H regimes as attractive regimes of operation for a fusion device. However, to fully realize this potential, it is necessary both to sustain high fusion performance, and to demonstrate consistent core-edge solutions including a high density near the separatrix, and strongly dissipative divertor.

An important challenge for sustainment of high confinement states is the control of density and impurity accumulation. On DIII-D, small $n=3$ magnetic perturbations produced by internal coils (“l-coil”) have enabled stationary pressure and density without impurity accumulation, enabling high performance ($\beta_p$ ~2.9, $H_{95}$ ~1.6, $T_e$ ~0.2s, W ~1.9MJ) for the programmed duration of the shot. Fig. 7 compares otherwise similar $I_p$ = 1.6MA DIII-D shots with (red) and without (black) an $n=3$ perturbation applied from 2.3-4.4s. By employing the $n=3$ perturbation and a target value of $\beta_p$ ~3 in the NBI power feedback algorithm), uncontrolled density accumulation and onset of core tearing modes are avoided, enabling sustained operation. While fusion performance is substantially reduced from the peak values discussed in the previous section (in part due to ion-electron equilibration and reduction of $T_e,0$ from max values of ~16 keV to stationary values ~8 keV) it remains high, with sustained values of $<p>/T_e$ ~12 kPa s, $n_0 T_e$ ~0.9 $10^{20}$ m$^{-3}$ keV s, and $Q_{\text{DT,eq}}$ ~0.14. The drop in confinement time from the peak values to the stationary state appears to primarily result from ion-electron equilibration, reduced rotation, and in some cases core tearing mode onset. Detailed study of core transport is in progress.

Because the pedestal in the SH and NSH regimes is predicted to be limited by current-driven kink-peeling modes (see eg. red and blue lines in Fig 1b, 2a, 2b, the near-linear increase in predicted pedestal pressure with pedestal density indicates a pedestal limited by current-driven modes), it is expected that increasing the density near the separatrix will not negatively impact pedestal stability or pedestal pressure. This is in contrast to predictions and observations in cases where the pedestal is limited by pressure driven modes (see eg. the rightmost black lines in Fig 1b, 2a, 2b), where the pedestal pressure is predicted to degrade with density (and can degrade even further if resistivity becomes high enough to strongly drive resistive ballooning modes). This degradation of pedestal pressure with density can be particularly problematic in metal wall machines, which often use strong gas puffing to protect metal surfaces from erosion [6,28]. Along the SH/NSH branch, it is predicted that solutions with both high pedestal pressure and high density, including high separatrix density, are realizable and may make attractive core-edge operational regimes.

To test this hypothesis, a scan of $D_2$ gas rate has been conducted in a series of DIII-D discharges with $I_p$ ~1.95MA, $B_t$ ~2.17T, $\delta$ ~0.6, $Q_{\text{edge}}$ ~3.8. Fig. 8 compares cases with $D_2$ gas rates (lower traces in Fig 8a) of 3, 50, 70 and 110 torr L/s, applied from t=2.8-4s. Applying these levels of gas puffing does not significantly impact either $T_e$ ~0.14s (Fig 8a, upper traces), or $p_{\text{ped}}$ (Fig 8b). The pedestal density in these cases is held approximately constant at ~7 $10^{19}$ m$^{-3}$ (Fig 8c, upper traces) by employing feedback control of line average pressure.
density by varying the i-coil current (Fig 8c, lower traces). However, the separatrix density (Fig 8d) increases significantly from \(2.5 \times 10^{19} \text{ m}^{-3}\) to \(4 \times 10^{19} \text{ m}^{-3}\) as the gas rate is increased. Note that in the high fuelling case, both the pedestal and separatrix density are in the range of ITER design values [29], while no significant degradation of either core confinement or pedestal pressure is observed. The strike point \(T_e\) at the outer divertor plate is reduced by a factor of \(2\) in the high fuelling case (from \(45 \text{ eV}\) to \(22 \text{ eV}\)), and the strike point density increases by a similar factor, reaching values \(\sim 7 \times 10^{19} \text{ m}^{-3}\) in the high fuelling case.

Introducing a radiative impurity, such as \(N_2\), can further improve divertor performance. In DIII-D 177018, a combination of 37 torr L/s of \(D_2\) gas, and feedback controlled \(N_2\) injection, with a target of 5MW of divertor radiated power, is used to enhance radiative losses in the divertor. This combination leads to substantial reduction of \(T_e\) and enhancement of \(n_e\) near the outer strike point, as shown in Fig 9. The divertor radiated power reaches its target value of \(\sim 5 \text{ MW}\), while total radiative losses are \(\sim 7.5 \text{ MW}\) (out of a total injected power of 12 MW), with no appreciable degradation of pedestal pressure or confinement time. Much larger values of \(N_2\) injection do lead to degradation as radiative losses become very large.

4. DISCUSSION AND FUTURE WORK

A set of experiments on Alcator C-Mod and DIII-D, guided by theoretical predictions of the Super H (and Near Super H) regime, have achieved high pedestal pressure and global fusion performance. Pedestal pressure up to \(\sim 80 \text{ kPa}\) has been achieved on C-mod at toroidal and poloidal field near the ITER value, extending tests of the EPED model nearly to the ITER predicted \(p_{\text{ped}}\). High stored energy and peak fusion performance have been achieved on DIII-D, including DT equivalent fusion gain \(Q_{\text{DT,eq}} \sim 0.5\) and \(<p>T_e \sim 65 \text{ kPa s}\), at modest current (1.95MA), field (2.17T) and size (a\(\sim 0.6\text{m}\), \(V\sim 20\text{m}^3\)).

By employing \(n=3\) magnetic perturbations to control density and impurity accumulation, sustained operation at high performance (\(\beta_n \sim 2.9\), \(H_9 \sim 1.6\), \(\tau_c \sim 0.2\text{s}\), \(W\sim 1.9\text{MJ}\)) has been achieved on DIII-D for the full hardware-limited duration of the discharge (\(\sim 2.5\text{s}\)). Divertor temperature has been reduced, and divertor radiated power increased, via puffing of \(D_2\) and \(N_2\) gas, without significant degradation of \(p_{\text{ped}}\) or core confinement.

In addition to C-Mod and DIII-D, SH/NSH operation is predicted to be possible on ITER (Fig 10a), as well as in strongly shaped discharges in JET and JT-60SA. A full assessment of predicted SH/NSH performance requires coupled core-pedestal simulation [11-14]. It is also useful to consider simpler performance metrics to compare regimes. The expression for fusion gain, \(Q = P_{\text{aux}}/P_{\alpha}\), or similarly \(Q^* = P_{\text{aux}}/(P_{\text{aux}} + P_{\alpha})\), can be simplified by writing the fusion power in terms of a peaking, fusion reaction energy and \(T_1\) factor \((f_p)\); \(P_{\text{aux}} = f_p \langle p \rangle^2 \text{ V}\). The required auxiliary heating power, together with alpha heating power, \(P_{\alpha} + P_{\alpha}\), can then be cast in terms of a multiplier \((f_{\text{HL}})\) of a simplified LH transition power scaling, \(P_{\text{aux}} + P_{\alpha} = f_{\text{HL}} c_{\text{HL}} n_e B_t S\) (where \(S\) is
endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of aut

\[ Q^+ \sim f_q I_p \text{ A} n, \text{ where } f_q \text{ is an overall “quality factor” which scales with } f_q f_q^2 \beta_{N,ped}^{-2} / f_{GW} f_{LH}. \]

For example, for ITER to achieve Q=10 (Q=10/3), at \( I_p=15\text{MA}, B_0=5.3\text{T}, a=2\text{m}, \) requires a quality factor \( f_q=0.021 \) (in units of T\(^{-1}\) MA\(^{-1}\) m\(^2\)). DIII-D Super H experiments have achieved significantly higher values of \( f_q \) both peak (~0.15, or ~0.1 with only thermal fusion) and sustained (~0.07, or ~0.03 with only thermal fusion), with important caveats including that these discharges include strong co-torque, and do not suppress ELMs. Because Q is highly sensitive to \( T_e \), it is valuable to consider the closely related metric, \( <p>W/P_{heat} I_p \text{ A} n, \text{ which enables comparisons of discharges with a wide range of } T_e. \) As shown in Fig 10b, Super H experiments on DIII-D and C-Mod have achieved high values of this metric, sufficient for high performance on ITER at \( I_p < 15 \text{ MA}. \)

There are numerous important directions for future exploration of the Super H mode regime, including a) ELM suppression in co-injected discharges, b) compatibility with full divertor detachment, c) compatibility with low torque, d) understanding of tearing mode onset, e) resistive wall mode physics, f) exploration of high-Z transport, g) extension to additional devices, and h) higher bootstrap fraction. The high peak performance, sustainment of high performance, and compatibility with high density separatrix and divertor, all suggest that the SH/NSH regimes hold promise for attaining high fusion performance on ITER, as well as potentially enabling compact cost-effective pilot plant/DEMO designs.

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